Development of a Method for Comparison of Lightbridge's Advanced Fuel Materials Against Conventional UO_2 Fuel Performance

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ABSTRACT

Lightbridge Fuel™ is an advanced metallic nuclear fuel designed for use in water-cooled reactors, including existing and new build nuclear power plants, and small modular reactor designs. This paper summarizes a method to quantitatively compare advanced fuel materials and design performance using the OECD/NEA benchmarks of Three Mile Island Unit 1 (TMI-1) Main Steam Line Break (MSLB).

The OECD/NEA Pressurized Water Reactor (PWR) MSLB benchmark provides an opportunity to demonstrate the proposed method, serving to baseline PWR transient analysis of differing fuel materials in an open-source environment. Lightbridge used the TRAC/RELAP Advanced Computational Engine (TRACE) to model system transient response during MSLB, and the Purdue Advanced Reactor Core Simulator (PARCS) code to model TMI-1 cycle 1 & 2 steady-state core as well as TMI-1 all-metal fuel core designs. This evaluation compares results of TMI MSLB transient behaviors between outputs from original benchmark specifications, perturbation with Lightbridge methods generated TMI-1 Cycle 2 core design point kinetics, and perturbation with Lightbridge methods generated TMI-1 with advanced metallic fuel point kinetics.

Results from this method provide early insights into anticipated impacts to transient responses and reactors with introduction of Lightbridge fuel designs to PWRs. OECD/NEA benchmarks of Peach Bottom Unit 2 Turbine Trip provide future opportunities to expand this method to compare advanced fuel material performance in Boiling Water Reactors (BWR).

Keywords: Advanced Fuels, Accident Tolerant Fuels, Transient Analysis

1. INTRODUCTION

Lightbridge Fuel™ is an advanced, all-metal fuel (AMF) design for water cooled reactors. The design features multi-lobed helically twisted fuel rod geometries and seeks to provide improved fuel performance and accident tolerance[1]. A common challenge for advanced fuel concepts during design considerations is to develop an independently verifiable basis of comparison between existing fuel designs and newly proposed designs with differing materials, geometries, and operational requirements. The "Main Steam Line Break (MSLB) Benchmark" [2], [3] compiled by the US NRC and OECD NEA provides well-studied independently accessible specifications to model a MSLB transient. This paper summarizes a methodology to provide early fuel material performance insights, using the NEA MSLB Benchmark as a baseline to demonstrate the proposed method of assessment.

2. EVALUATION METHODOLOGY

In order to compare transient performance of Lightbridge Fuel to standard UO_2 fuel, a set of consistent systems transient analysis input decks using benchmark thermal hydraulic, point kinetics neutronic, and reactor protection system specifications is necessary. For this exercise, Lightbridge selected the TMI Unit 1 MSLB benchmark [2], [3]. The evaluation methodology consists of three basic parts, as described below.

First, as a baseline, the TMI Unit 1 MSLB benchmark transient was simulated and compared to original benchmark results, providing software quality assurance and transient model input validation for the purpose of this assessment. Next, since core designs are not provided as part of the TMI Unit 1 MSLB benchmark data, a reference oxide fuel core (ROFC) must be created. The MSLB transient was therefore simulated using inputs which were modified with point kinetics inputs derived from Lightbridge-developed core design methods for the TMI Unit 1 Cycle 1 & 2 benchmark cores using standard UO_2 fuel. These ROFC results provide insights into impact from selected core design codes and methods prior to introduction of proposed new fuel materials. Finally, Lightbridge AMF material properties and fuel-rod interior geometries are inputted into both core design and transient analysis models, and the MSLB transient is simulated without further modifications.

The results between the baseline, ROFC, and AMF simulations are compared for final assessments of proposed fuel material performance to provide early insights. The method is illustrated in Figure 1, with demonstration selections in parenthesis, including the codes that Lightbridge selected for this demonstration. Details of selected codes and demonstration models are further discussed in Sections 2.1 through 2.3.

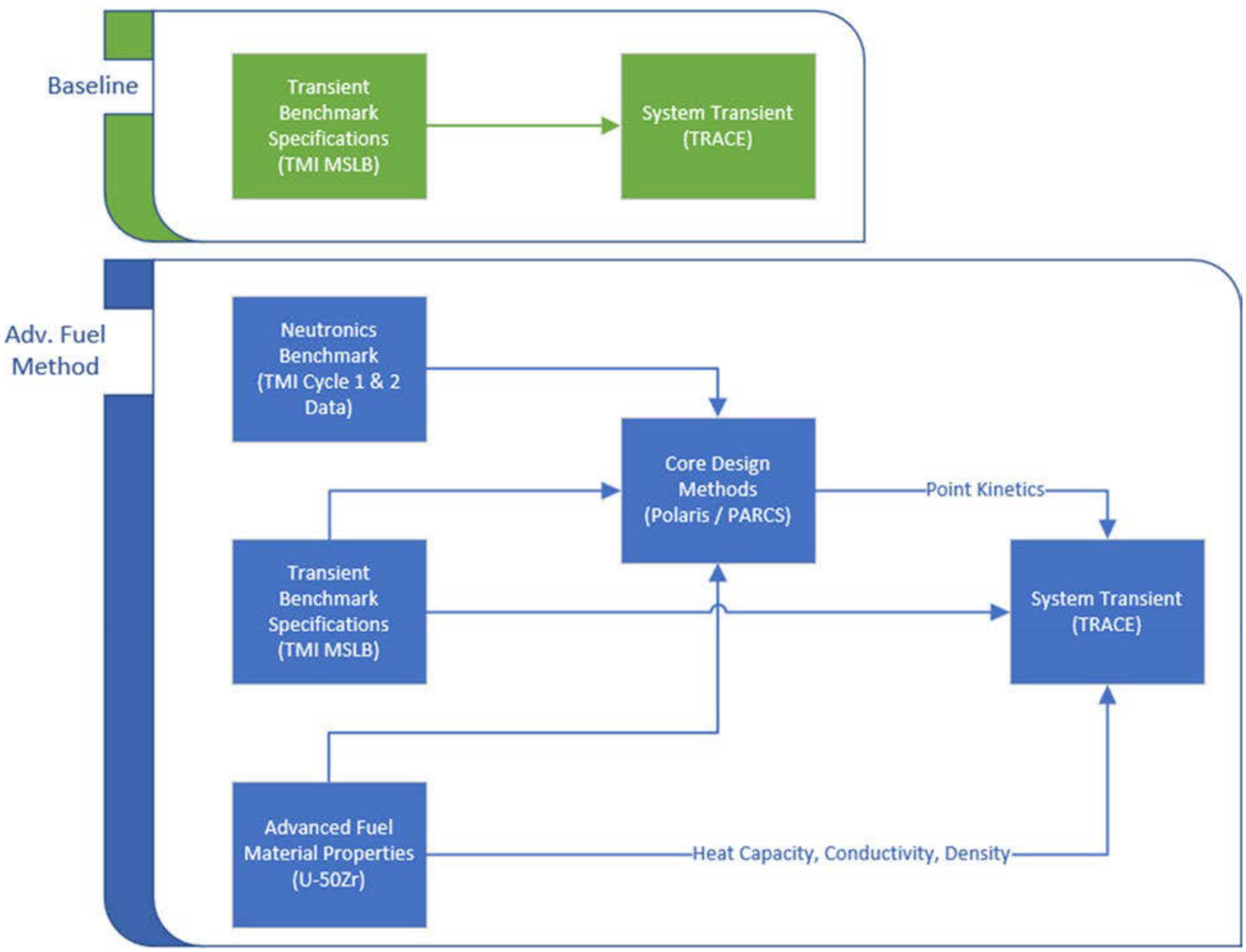

Figure 1. Benchmark & Evaluation Method Flow Diagram

2.1. TMI Cycles 1 & 2 Reference Core Designs with Oxide and All-Metal Fuels

The reference oxide fuel core design for the NEA MSLB Benchmark is based upon the reactor geometry and operational data from TMI Cycles 1 and 2 [3]. TMI core models for neutronics analysis were developed using the Polaris and PARCS nuclear analysis codes from core design and operational data from the first two TMI cycles [4]. Polaris is a 2-dimensional lattice physics code, developed specifically for LWR analysis to perform assembly lattice depletion and generate few-group cross-section data for nodal core simulator codes, such as PARCS. PARCS is a 3-dimensional reactor core simulator which solves the steady-state and time dependent multi-group neutron diffusion equation. The purpose of the neutronics analysis is to calculate point kinetics parameters, using Lightbridge core design methods, for use in the transient analysis of the MSLB problem to evaluate U-Zr fuel performance against standard UO_2 fuel.

The assembly lattice design modeled in Polaris for the nominal TMI core with UO_2 fuel is a 15x15 layout containing 208 fuel pins, 16 guide tubes and 1 instrument tube. The fuel pin pitch is 1.443 cm, and the active fuel length is 365.76 cm. The fuel pellet radius is 0.470 cm, and the clad outer radius is 0.546 cm with a thickness of 0.067 cm. The cladding is composed of Zircaloy-4 and the gas gap is modeled as helium filled. The guide tubes and instrument tubes are composed of Zircaloy-4 with an outer radius of 0.626 cm and thickness of 0.066 cm.

For U-Zr fuel, the assembly lattice retains the same dimensions and components as the standard fuel model within the 15x15 assembly layout. The fuel form remains cylindrical; the outside radius of the fuel rod remains the same, as does the composition and dimensions of the cladding. The fuel is composed of approximately 50% uranium and 50% zirconium by weight. The fuel rod contains a central displacer which was modeled as a mixture of zirconium with 10 wt% to 30 wt% erbia (Er_2O_3) for reactivity control. The central displacer forms about 10% of the fuel volume. The U-Zr fuel extends from the displacer to the cladding with no gap.

The TMI core incorporated full-length and part-length control rod assemblies (CRAs). The absorber material for both was silver-indium-cadmium (Ag-In-Cd), and the cladding was 304 stainless steel. TMI Cycle 1 used burnable poison rods (BPRs) composed of Al_2O_3-B_4C within the guide tubes of several assemblies. The BPRs were removed for Cycle 2, and no BPRs were used within the U-Zr fuel assemblies.

The 56 group-neutron 19 group-gamma nuclear data library based on ENF/B-VII was used for the generation of cross-sections in Polaris. Cross-sections were generated for four fuel enrichments (2.06, 2.75, 3.05, and 2.64 wt% U^{235}) with up to three BPR loadings (1.43, 1.26, and 1.09 wt% B_4C) for a total of eight different UO_2 fuel assemblies used in the TMI core for Cycles 1 and 2. For U-Zr fuel assemblies, four fuel enrichments ranging from 5.0 to 8.0 wt% U^{235} were used for Cycle 1, and four fuel enrichments for the fresh fuel ranging from 11.0 to 15.0 wt% U^{235} were needed for Cycle 2. The amount of erbia in the central displacer ranged from 10 wt% to 30 wt%.

The PARCS code was used to model the TMI core with UO_2 fuel based on the actual loading pattern designs from TMI Cycles 1 and 2. The core contained 177 assemblies surrounded by an 18-inch radial reflector. Approximately 1/3rd of the core is loaded with each of the first three UO_2 fuel batches with the highest enriched fuel located on the core periphery, and the remaining two fuel batches arranged in checkerboard pattern within the core. There were 68 BPR assemblies placed in core locations without CRAs. The second cycle contained the fourth fuel batch of 56 fresh assemblies which were placed primarily on the periphery of the core. No BPRs were loaded for Cycle 2.

For U-Zr fuel, the PARCS core models for Cycles 1 and 2 were designed to match the reactivity of those cycles with UO_2 fuel, which was done so that differences in the point kinetics results would be based primarily on the properties of the different fuel compositions, not on the properties associated with different core designs. For Cycle 1, the highest enriched fuel was placed on the periphery of the core with the other batches mixed within the core interior. The reload batch size of 56 fresh assemblies for Cycle 2 is the same as the reload batch size for the UO_2 fuel, but five different assembly types with varying enrichments and erbia loadings were needed to match the reactivity of the UO_2 design.

The operating conditions input to the TMI core models were a core rated thermal power of 2535 MW_{th}, an inlet temperature of 290 ºC (554 ºF), an average coolant temperature of 304.3 ºC (579.7 ºF), and the core pressure was 2200 psi. Soluble boron was used to control excess reactivity. The core models were depleted to the end-of-cycle (EOC) at 100% power with all CRAs withdrawn from the core resulting in cycle lengths of 550 effective full power days (EFPD) for Cycle 1, and 290 EFPD for Cycle 2 with UO_2 fuel, and 559 EFPD for Cycle 1 and 290 EFPD for Cycle 2 with U-Zr fuel. Point kinetics parameters were calculated for Cycle 2 EOC for both fuel composition models, including moderator temperature coefficient (MTC), Doppler temperature coefficient (DTC), total CRA worth, and delayed neutron fraction. The reduced U^{238} content within the metallic fuel reduces the magnitude of the DTC and the MTC, and increases the delayed neutron fraction. The total CRA worth is reduced for the metallic fuel primarily due to the loading pattern design. However, the total CRA worth calculated for the Cycle 2 core models, for both fuel compositions, is greater than what was provided in the NEA MSLB Benchmark [3]

Note that TMI Cycles 1 and 2 were operated at a rated thermal power of 2535 MW_{th}. The NEA MSLB Benchmark [3] specified a rated power of 2772 MW_{th}. This difference in power does not appreciably change the differences in the calculated kinetics parameters between UO_2 and U-Zr fuel.

2.2. U-50Zr Fuel Thermophysical Properties

Lightbridge fuel is composed of a binary U-Zr phase, with the zirconium content placing the alloy within the delta phase (42-60 wt. %). The majority of material property information in literature primarily covers the alpha phase (U-10Zr) from the legacy metallic fuels programs, such as experimental breeder reactor (EBR) II and the fast flux test facility (FFTF), with limited measurements being characterized within the delta phase. INL characterized material properties of U-50Zr in [5] and further characterizations of Lightbridge's specific alloy are being conducted. Within this work, the measurements supplied by INL were used within this analysis and are summarized within this section due to the material composition best representing Lightbridge's fuel form. These thermal properties were supplied to the PARCS and TRACE analyses, with plans to use them within fuel performance scoping studies.

For specific heat capacity, measurements were taken from as-cast U-50Zr and were taken upon heating. During the heating process, two phase transitions were observed, and the measurements within the transition zones excluded. Thermal conductivity and linear thermal expansion measurements were obtained from the same as-cast microstructure within the INL characterization report [5].

The thermal material properties for the U-50Zr were digitized and formatted for the PARCS and TRACE advanced fuel material input models. The properties for the Zircaloy cladding and the benchmark UO2 fuel used within the analyses are based off the MATPRO data found within the PARCS code. These material properties are obtained over decades of experiments and validation within LWR reactors, both for Zr4 and Zr2 alloys [6], [7].

2.3. NEA/OECD TMI MSLB Transient Analysis Model

TRACE is an NRC thermal-hydraulics code able to analyze large/small break LOCAs and system transients in both PWRs and BWRs. The TRACE TMI MSLB baseline model was developed from NRC/NEA MSLB benchmark specifications of TMI unit 1, a B&W designed pressurized water reactor (PWR). The TRACE input deck explicitly models 2 Steam Generators (SG), 4 Reactor Coolant Pumps (RCP), 2 main steam lines, and reactor hydraulic volumes. Additionally, all credited reactor protection systems (RPS) according to benchmark specifications are modeled including high flux trip, low pressure trip, and control rods. The model also includes high pressure Emergency Core Cooling (ECC) injection and main steam isolation on low Reactor Coolant System (RCS) pressure and low steam line pressure respectively.

For AMF runs, the TRACE MSLB input decks are modified to incorporate AMF fuel rod interior dimensions and material properties. The key differences between baseline/UO_2 and AMF fuel geometries are the removal of the fuel rod gas gap and replacement of a portion of fuel centerline material with Zircaloy material to represent fuel displacers. In order to model thermal hydraulics behavior within existing range of validation, the fuel rod outer geometry and fuel assembly dimensions remain unchanged for this demonstration. Details of changes and differences between UO_2 and AMF fuel models are further discussed in Section 2.1.

The model is initialized at hot full power conditions for all 3 cases. The MSLB transient begins with a double guillotine 24-inch break and a simultaneous 8-inch limiting velocity break per the benchmark specifications.

2.4. Transient Analysis Results

The results between baseline, ROFC, and AMF are compared for performance insights. For all three transient runs, the MSLB transient begins with a double guillotine 24-inch break and a simultaneous 8-inch limiting velocity break per the benchmark specifications.

The at-fault SG rapidly depressurizes, causing overcooling in the reactor system. Overcooling of the RCS results in a positive reactivity and consequently power increase response due to densification of the moderator. The main steam isolation valves close on low main steam line pressure. Main feedwater isolation/regulating valve remains stuck open and aligned with the at-fault SG for a prespecified duration and flow rates, in conformance with the MSLB benchmark specification [3]. RPS actuation occurs due to high flux, with full insertion of control rods after 2.3 seconds. Post SCRAM, total reactor power falls quickly to decay heat load. The main steam relief valves lift open on the isolated in-tact steam generator due to over-pressurization.

As the transient progresses from initial rapid overcooling to decay heat removal, both SG continue to remove decay heat load, with a decrease in feedwater flow due to specified boundary condition 10 seconds after start of transient. This reduction resulted in a brief mismatch between decay heat load and removal, resulting in a brief period of overheating observed in all parameters of interest. Specified main feedwater flow to the at-fault SG terminates after 42 seconds; however, ECC injects on low RCS pressure within the same period. The continued cool down of the reactor during the latter portion of transient could result in the core observing return to criticality due to negative moderator feedback. However, Lightbridge modeled core designs predict significantly larger shutdown margin and less negative moderator feedback coefficient for both ROFC and AMF cases in comparison to the original MSLB benchmark specifications. Prior to specified termination of the transient at 100 seconds, the at-fault steam generator experiences dry out conditions. This decreases heat removal; thus, increasing RCS temperatures and resulting in negative moderator reactivity feedback.

Table I. Demonstration Sequence of Events

Time (sec) ROFC	Time (sec) AMF	Event	Value
0.0	0.0	Steam Line Break	-
6.02	6.69	MSIV Isolation on Low Pressure	[4.24 MPa] / (615 psia)
6.44	7.10	Reactor Trip on High Flux	114% RTP
6.87	7.51	MSRV Lift	[7.35 MPa] / (1067 psia)
35.34	29.36	MSRV Close	-
44.34	40.94	ECCS on Low RCS Pressure	[11.34 MPa] / (1645 psia)
100	100	Benchmark Terminated	-

The baseline. ROFC, and AMF transient results are plotted in Figure 2, Figure 3, Figure 4, and Figure 5 along with the original benchmark results. The parameters of interest plotted are total reactor power, total core reactivity response, average fuel temperature, and average core coolant temperature. Typical MSLB transient analysis assesses core reload performance for Departure from Nucleate Boiling (DNB), peak centerline temperature, and if the limiting core design has sufficient shutdown margin to prevent re-criticality. However, given the goals and geometric limitations of this evaluation method, the presented set of parameters allows fuel designers and independent parties to assess anticipated material performance.

The baseline results are consistent with the benchmark results, indicating that TRACE code and transient models are sufficient for this assessment. The ROFC results show the impact of the core design codes and methods utilized by Lightbridge in this work for oxide fuel in comparison to the baseline, providing a set of results that may be used to compare to the AMF results. Both ROFC and AMF results show increased shutdown margin with no return-to-power due to both an increase in total SCRAM worth and less negative MTC.

Based on a comparison of the ROFC and AMF results, the U-50Zr fuel material anticipated performance is within expectations. With lower initial and transient fuel temperatures, the all-metal fuel is predicted to maintain sufficient margins to any potential temperature design limits based on fuel material properties. RCS temperature response to power transients is faster with lower heat holdup and less insulation due to the lack of a gas gap and lower specific heat capacity. However, the TMI MSLB benchmark doesn't provide a period of loss of RCS flow or prolonged overheating. Therefore, further demonstrations focusing on at-power overheating events such as PWR locked rotor and BWR turbine trip merit analysis to assess advanced fuel material performance during heat up and dry-out conditions. Demonstrations focusing on rapid energy depositions such as rod eject accidents also merit analysis to assess proposed material performance during rapid localized power excursion and subsequent energy deposition.

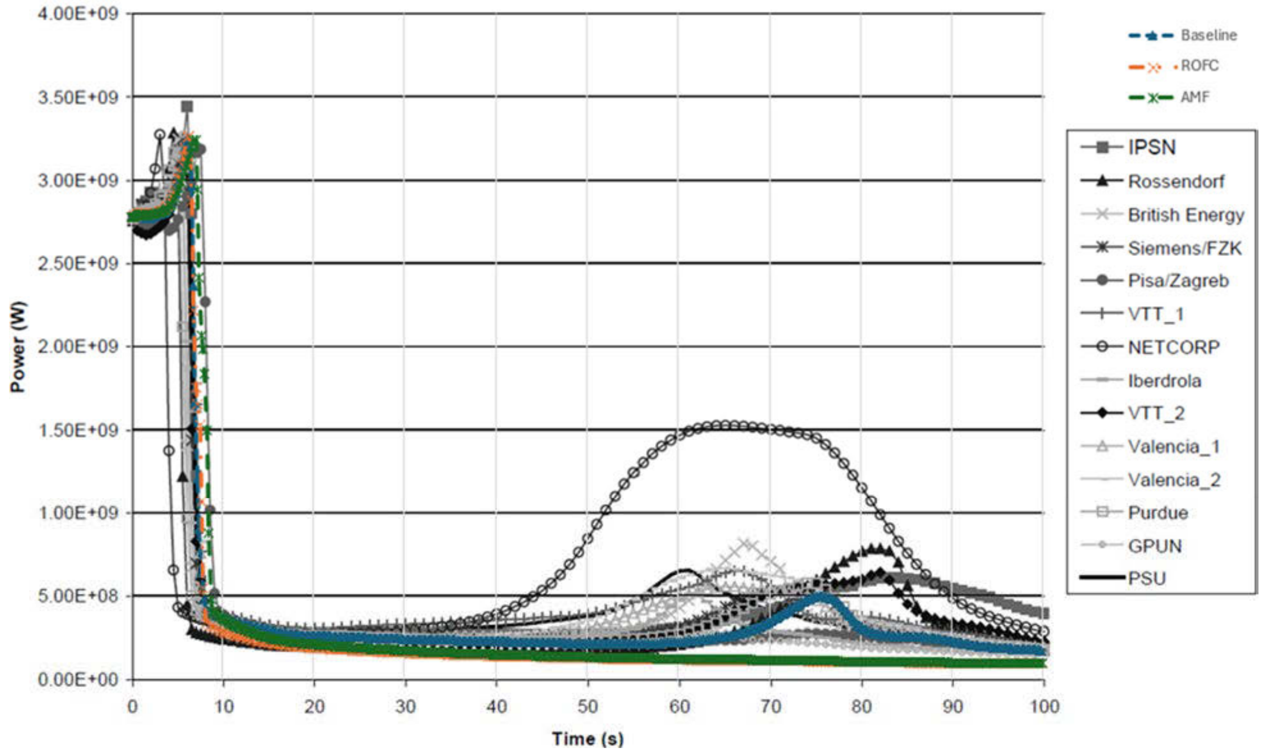

Figure 2. Reactor Power Response Benchmark [2] **vs Demonstrations**

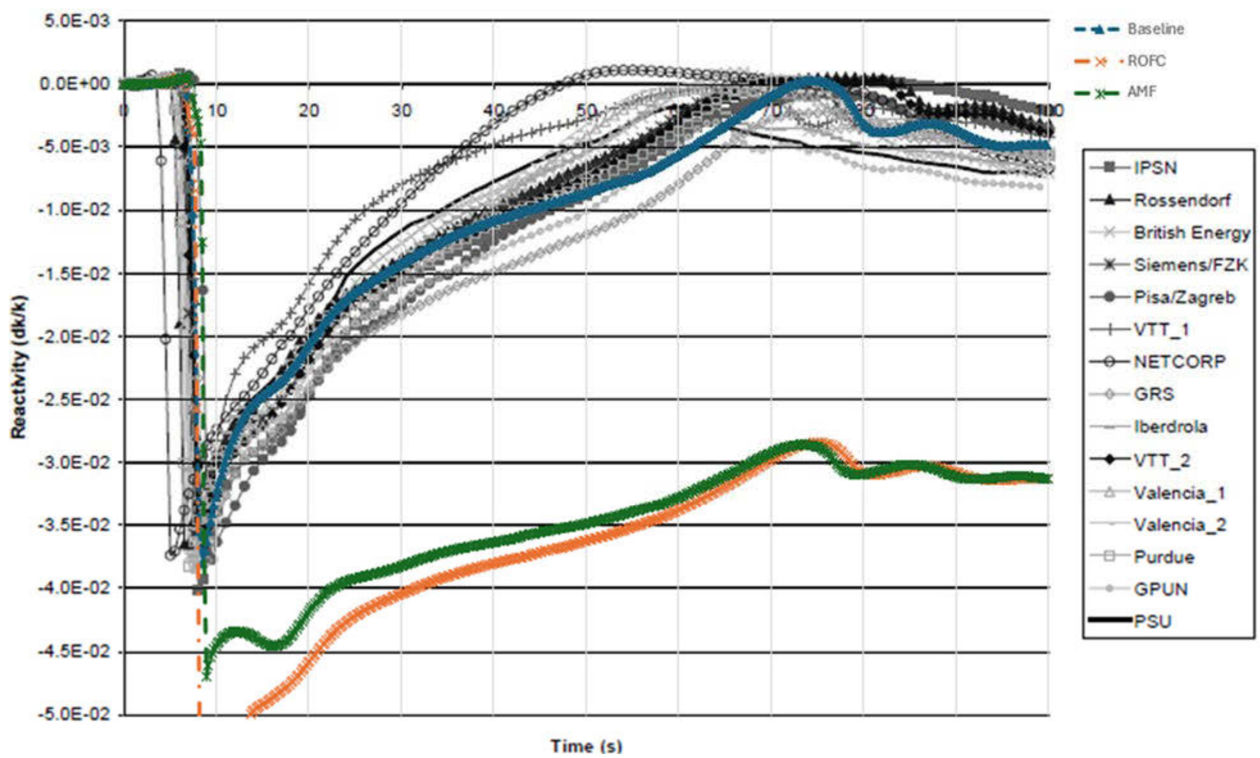

Figure 3. Reactivity Response Benchmark [2] **vs Demonstrations**

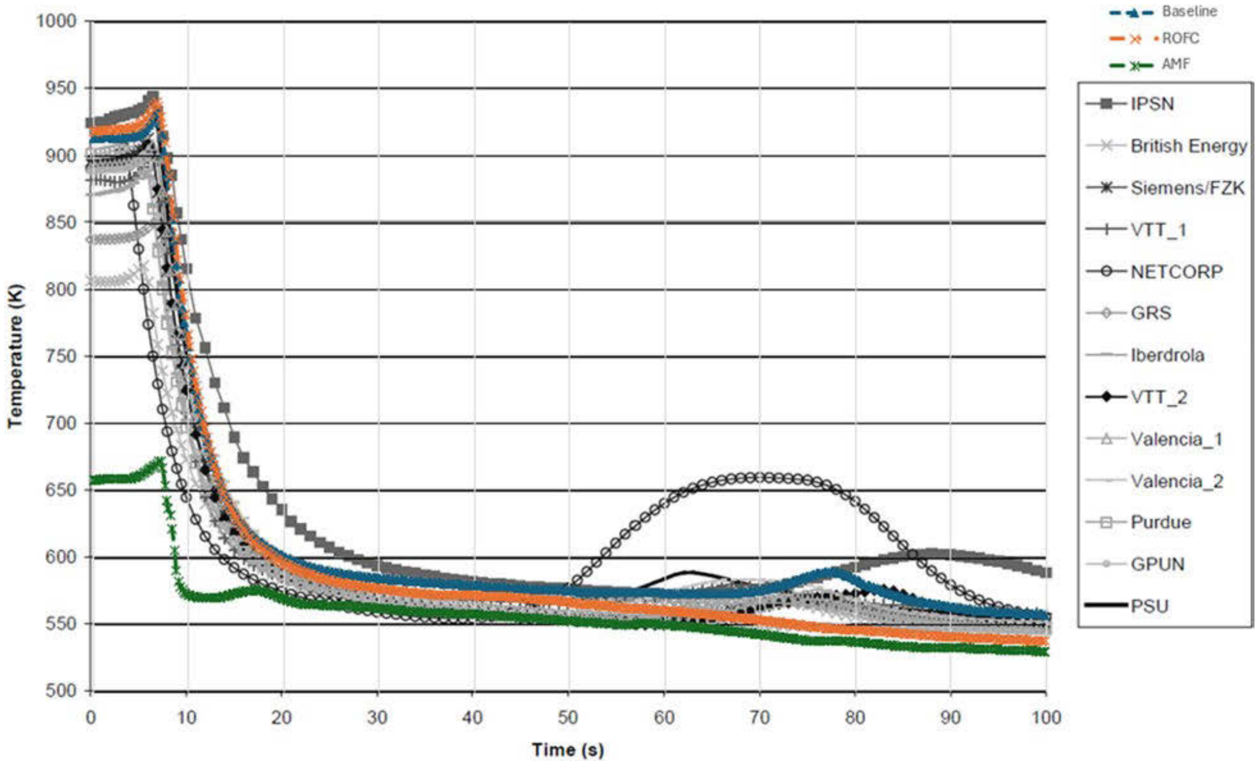

Figure 4. Average Fuel Temperatures Benchmark [2] vs Demonstrations

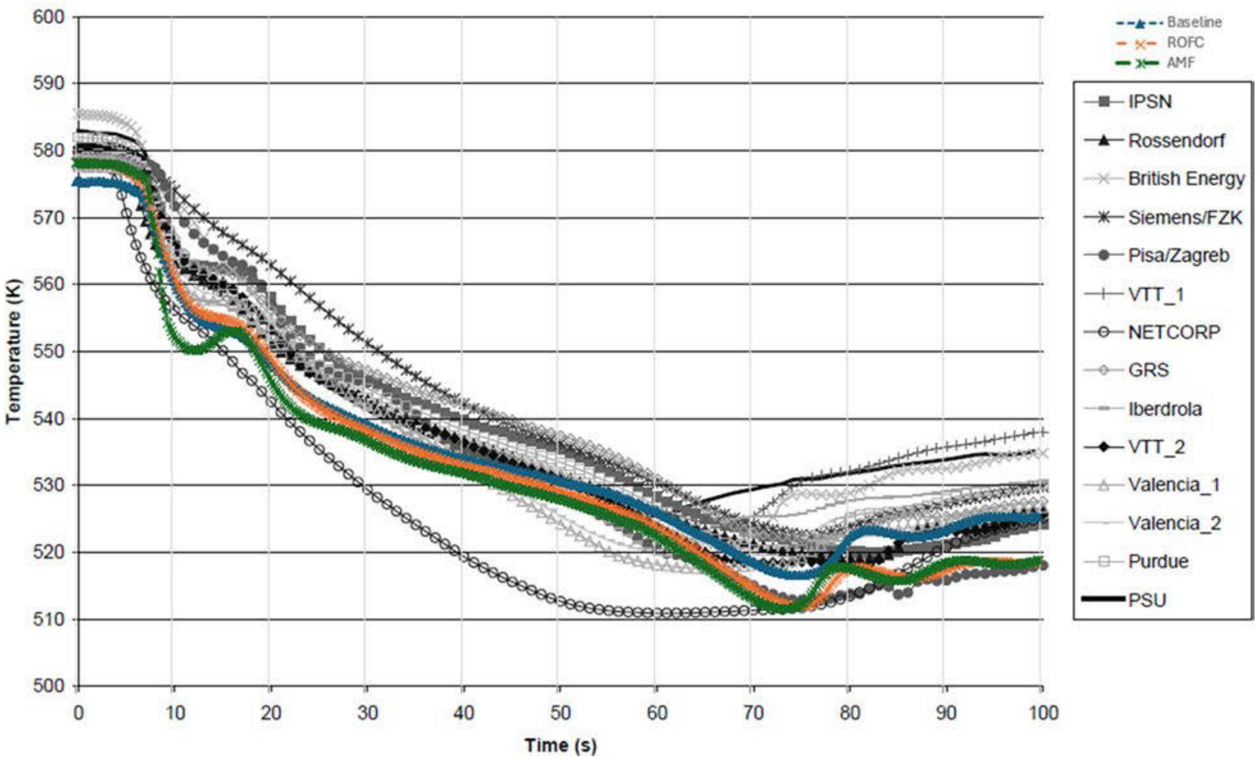

Figure 5. Average Core Coolant Temperatures Benchmark [2] vs Demonstrations

3. CONCLUSIONS

The proposed advanced fuel material evaluation method was exercised through the demonstration of Lightbridge Fuel™ materials experiencing a TMI MSLB transient originally benchmarked by OCED/NRC and the results indicate that the method presented in this work is suitable for comparing Lightbridge AMF transient performance to that of standard oxide fuel.

The TMI MSLB point kinetics inputs were calculated from core design models of TMI Unit 1 Cycles 1 & 2 based on TMI operating data documented by EPRI[4]. AMF material properties were incorporated from INL's characterization of U-50Zr. The results of the demonstration showed expected advanced material performance from U-50Zr, driven by increased thermal conductivity and removal of the gas gap. The U-50Zr fuel operates at and reaches lower maximum fuel temperatures during the transient, providing a significant margin of safety for peak fuel temperatures.

The predicted significant margin to peak fuel temperatures warrants the development of analyses to study Lightbridge Fuel™ performance in PWRs operating at extended uprate conditions with longer fuel cycles. The observed results also merit further exercises of the method to assess advanced fuel material performance during rapid and long-term heat up events. Additionally, the method could be applied to other reactor technologies that are well benchmarked, including Boiling Water Reactors, and CANDU heavy-water reactors. These efforts will further guide Lightbridge fuel design considerations to support additional applications.

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